NON-PUBLIC?: N
ACCESSION #: 9301250006
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Catawba Nuclear Station, Unit 2 PAGE: 1 OF 10

DOCKET NUMBER: 05000414

TITLE: Main Feedwater Pump Trip During Testing and Subsequent
Reactor Trip
EVENT DATE: 12/14/92 LER #: 92-006-00 REPORT DATE: 01/13/93

OTHER FACILITIES INVOLVED: N/A DOCKET NO: 05000

OPERATING MODE: 1 POWER LEVEL: 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR
SECTION:
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
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COMPONENT FAILURE DESCRIPTION:
CAUSE: SYSTEM: COMPONENT: MANUFACTURER:
REPORTABLE NPRDS:

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On December 14, 1992 at 1533 hours with Unit 2 in Mode 1, Power Operation, a reactor trip occurred due to low-low level in Steam Generator (S/G) 2D. The reactor trip occurred after the 2A main feedwater pump (CFPT 2A) tripped during the weekly test of the pump's overspeed trip mechanism. The subsequent investigation determined that the momentary relaxation of the operator's hand while holding the trip lockout switch in the lockout position caused the pump to trip. The root cause of the CFPT 2A trip is attributed to its inadequate testability features. The testability features of the CFPTs and the test procedure are being evaluated to identify possible improvements to increase the reliability of the CFPT test. The Digital Feedwater System (DFS) initiated a runback but was unable to prevent the reactor from tripping. The initial investigation of the DFS did not identify any problems, therefore the root cause is unknown. Further investigation is being conducted to identify the root cause. The only significant anomalous

behavior identified after the reactor trip was the Reactor Coolant (NC) System loop B average temperature (T-ave) indication. The T-ave indication caused the steam dump valves to open, but were closed only seconds after opening by the low T-ave P-12 interlock when the NC system temperature reached 553 degrees F. Process control cards were determined to be the cause of the T-ave indication problem and have been replaced.

END OF ABSTRACT

BACKGROUND

The Main Feedwater EIIS:SJ! (CF) System consists of two steam generator feedwater pumps EIIS:P!, two stages of high pressure feedwater heaters (A and B), piping EIIS:PSP!, valves EIIS:V!, and instrumentation. Normally, both feedwater pumps will be operating with each pump handling half the feedwater flow. Downstream of the feedwater pumps, the feedwater passes through two stages of high pressure heaters to a final header where the temperature is equalized. The feedwater is then admitted to the steam generators EIIS:HX! (S/G) through four steam generator feedwater lines, each of which contains a control valve and a flow nozzle.

The Digital Feedwater System (DFS) is composed of various controllers EIIS:XC! and equipment whose primary function is to maintain a programmed level in each S/G. In addition to steady state controller function, the DFS also restores and maintains S/G level to an acceptable value during normal plant transients and therefore preventing a reactor trip. Feedwater flow is controlled by control valves and pump speed. The control valves are positioned by a three element feedwater control system using feedwater flow, steam generator water level, and main steam flow as control parameters. The main feedwater pumps' speed is adjusted based on a programmed differential pressure (D/P) between the main steam header and the feedwater discharge header. The main feedwater pump speed is operated in conjunction with the feedwater control valves to maintain S/G water level while maintaining optimum valve position.

The purpose of the feedwater isolation signal is to initiate isolation of each steam generator and rapidly terminate feedwater flow and steam blowdown inside containment following a main steam or feedwater line break in containment EIIS:NH!, to prevent loss of steam generator water inventory due to a pipe rupture outside containment, and to prevent overfilling the steam generators if for some reason the normal means of controlling steam generator level malfunctions. Feedwater isolation is activated by any one of the following signals: safety injection, reactor trip plus low average reactor coolant temperature (T-ave=564 degrees F), or Hi-Hi Steam Generator level. A feedwater isolation signal closes the Feedwater Isolation Valves, Feedwater Purge Valves, Feedwater Control Valves, Feedwater Control Bypass Valves, Feedwater Preheater Bypass Valves, Feedwater Bypass Tempering Flow Valves, and Feedwater Pump Discharge Isolation Valves.

The Auxiliary Feedwater EIIS:BA! (CA) System assures sufficient feedwater supply to the steam generators in the event of loss of the CF System, to remove primary coolant stored and residual core energy. The system is designed to start automatically in the event of loss of offsite

electrical power, trip of both CF pumps, safety injection signal, or low-low steam generator water level; any of which may result in, coincide with, or be caused by a Reactor trip. In addition, the CA System will supply sufficient feedwater flow to maintain the Reactor at hot standby for two hours followed by cooldown of the Reactor Coolant EIIS:AB! (NC) System to the temperature at which the Residual Heat Removal EIIS:BP! (ND) System may be operated.

The CA System consists of three auxiliary feedwater pumps; each powered from separate and diverse power sources. Two full capacity motor EIIS:MO! driven pumps are each capable of supplying feedwater to two steam generators. These pumps will start automatically and provide the minimum required flow against a S/G pressure corresponding to the set pressure plus 3% accumulation of the lowest set main steam safety valve within one minute following initiation of the system. Initiation conditions are any one or combination of the following: 2 of 4 low-low level alarms in any 1 of 4 S/Gs, loss of all CF pumps, safety injection, or loss of offsite power. In addition, a turbine EIIS:TRB! driven pump is capable of supplying feedwater to two S/Gs. This pump is driven from steam contained in either of the two S/Gs. The turbine driven pump will start automatically and provide the minimum required flow against a S/G pressure corresponding to the set pressure plus 3% accumulation of the lowest set main steam safety valve. This pump will start automatically on either one of the following conditions: 2 of 4 low-low level alarms in any 2 of 4 S/Gs or loss of offsite power.

The purpose of the Steam Dump EIIS:JI! (IDE) System is to: 1) enable the Reactor to follow Main Turbine load reductions of less than 5%/min. ramp or 10% step change and 15%/min. ramp or 30% step change; 2) allow unit load reduction from 100% to plant auxiliary loads without a Reactor trip; 3) allow a Turbine trip and Reactor trip from 100% without lifting the Main Steam EIIS: SB! (SM) System Safety Valves. The system accomplishes its purpose by the use of five banks of dump valves divided into condenser dumps and atmospheric dumps. Condenser dump valves are divided into three banks with three valves per bank. Atmospheric dump valves are divided into two banks with four and five valves per bank, respectively. The total capacity of the Steam Dump System is 71.5% of the total unit capacity.

The condenser and atmospheric dump valves are controlled by one of three controllers EIIS:KC! (steam pressure, load rejection, plant trip). The selected controller actuates to control T-ave at or near a set reference signal. The reference signal to the dump valves is filtered through a pneumatic circuit which contains block valves and arming valves. This "block" circuit prevents cooldown below 553 degrees F to ensure T-ave remains above the minimum temperature for criticality.

The P- 12 (Lo Lo T-ave) Interlock is part of the Engineered Safety Features (ESF) Actuation System. The purpose of the interlock is to block steam dump valve actuation to prevent excessive cooldown below the minimum temperature for criticality. The setpoint for P-12 is 553 degrees F on any two of four NC System loops.

The purpose of the pressurizer EIIS:VSL! is to control the NC System pressure. In the pressurizer water and steam are maintained in equilibrium by electrical heaters EIIS:EHTR! and water spray. Steam can be formed (by the heaters) or condensed (by the spray) to reduce pressure variations due to contraction and expansion of the Reactor Coolant. The liquid level in the pressurizer is maintained by level transmitters EIIS:XT! which provide signals for use in the Reactor Coolant and Protection System (RCPS), the Safety Injection EIIS:BQ! (NI) System and the Chemical and Volume Control EIIS:CB! (NV) System. Each transmitter provides an independent high water level signal that is used to actuate an alarm and, upon 2 out of 3 coincident alarms, will cause a reactor trip. The transmitters also provide independent low water level signals that will activate an alarm. If the liquid level falls to a fixed low level alarm setpoint, it will trip the pressurizer heaters "off' and close the letdown line isolation valves.

The purpose of PT/2/B/4250/04A, Feedwater Pump Turbine Weekly Test, is to test the Overspeed Governor and Tripping Mechanism, Main Oil Pumps and Emergency Bearing Oil Pump, Thrust Bearing Wear Trip, and Hi and Lo Oil Tank Level Alarms. These tests are required per the General Electric Instruction Book, GEK-48666, CNM-1200.01-43.

The Overspeed Trip is tested by blocking the actual overspeed trip and hydraulically actuating the overspeed trip mechanism as indicated by the trip light.

EVENT DESCRIPTION

On December 14, 1992 Unit 2 was operating in Mode 1 at 100% Reactor power.

At approximately 1530 hours, Operations was performing procedure PT/2/B/4250/04A, Feedwater Pump Turbine Weekly Test, on CFPT 2A. Per step 12.2. 1.1 of the procedure the Operator turned and held the overspeed trip lockout switch in the lockout position, then per step 12.2.1.2 depressed the overspeed trip test push button. The Operator did not receive the expected "trip exercised" light. While holding the trip lockout switch in the lockout position, the Operator attempted to remove the light bulb. The momentary relaxation of the Operator's hand while holding the trip lockout switch in the lockout position caused the pump to trip at 15:31:19. The pre-trip reactor and turbine runback began.

CFPT 2B and the feedwater control valves functioned as designed to control feedwater flow. The condenser dumps valves functioned as designed to control T-ave. The pressurizer spray valves and heaters functioned as designed to control reactor coolant pressure. Control Room Operators (CROs) entered procedure AP/2/A/5500/03, Load Rejection.

At 15:33:41, the reactor tripped due to S/G 'D' lo lo level (38.6%)and the main turbine tripped due to the reactor trip. The reactor had runback to 83% power and the main turbine load had runback to 804MWe at the time of the trip. The CA motor driven pumps started as expected. Subsequently, S/G 'C' reached it's lo lo level setpoint and the CA turbine driven pump started as expected on 2/4 S/G lo lo level.

CROs immediately entered procedure EP/2/A/5000/01, Reactor Trip or Safety Injection, to verify the plant properly responded to the trip and to assess plant conditions. CROs then entered procedure EP/2/A/5000/01A, Reactor Trip Response, per EP/2/A/5000/01.

At 15:33:58, a CF isolation occurred as a result of the reactor trip with Low T-ave (
functioned properly.

At 15:34:27, NC Loop 'B' T-ave began an erratic trend upward.

At 15:34:33, Condenser Dump Valves began opening in response to the NC

Loop 'B' T-ave upward trend.

At 15:34:48, all Condenser Dump Valves closed due to P-12 (2/4 loops T-ave < 553 degrees F).

At 15:36:55, all pressurizer (PZR) heaters de-energized due to PZR levels decreasing to 17%.

At 15:36:58, NV Letdown isolation initiated due to PZR level reaching 17%. CROs immediately entered procedure AP/2/A/5500/12, Loss of Charging or Letdown. The minimum PZR level reached was 14.9%.

At 15:39:40, Operators throttled CA flow to control cooldown, valves 2SA2 (S/G 2B SM TO CAPT) and 2SA5 (S/G 2C SM TO CAPT) were closed. The minimum NC temperature reached was 530 degrees F. CROs defeated Delta-T and T-ave logic on NC 'B' loop.

At 15:41:29, Operators manually opened 2NV252 (NV PUMPS SUCT FROM FWST) and 2NV253 (NV PUMPS SUCT FROM FWST) in anticipation of an autoswap due to low Volume Control Tank (VCT) level.

At 15:42:50, PZR Heaters energized due to the recovering PZR levels.

At 15:51:03, NV Letdown was reestablished.

At 1630 hours, plant was stabilized in Mode 3, Hot Standby with NC temperature at 555.7 degrees F and NC pressure at 2241 PSIG.

At 1653 hours, the required notification for the reactor trip and ESF actuations were made to the NRC.

CONCLUSION

This incident was initiated by the loss of main feedwater pump CFPT 2A and not prevented by the Digital Feedwater System (DFS). The root cause of the CFPT 2A trip is the inadequate testability features of the CFPT overspeed governor and tripping mechanism. The root cause of the DFS's failure to prevent the reactor trip is unknown.

The subsequent investigation determined that the momentary relaxation of the Operator's hand while holding the lockout switch in the lockout position caused the pump to trip. The test requires that a spring loaded 'return to normal' pistol grip trip lockout switch be held in the lockout position while performing the test. The switch turns 45 degrees clockwise to the lockout position with the lockout activating at approximately 22 degrees. During past outages, while circuit calibrations were performed, IAE technicians have inadvertently deactivated the lockout by easing tension on the trip lockout switch. The technicians were completely unaware that the switch was moving until the lockout deactivated. Each time this happened the switch was being field in the lockout position for an extended period of time, the technician was working unassisted, and was distracted by an unanticipated system response. This deactivation of the lockout also occurred during this investigation while conducting a heat-up test on the lockout circuit.

The Operator performing the test is experienced at performing this test and was properly following the correct procedure. The Operator was aware that actions taken during this test, if not completed successfully, could trip the pump and have a significant impact on the operation

of the plant. The Operator was working unassisted, in accordance with the procedure, while performing the test and received an unanticipated response when the "trip exercised" light failed to illuminate. The Operator was holding the trip lockout switch in the lockout position for an extended period of time while trying to replace the "trip exercised" light bulb with the other hand. The Operator was unaware that the switch was moving until the pump tripped.

The trip lockout switch and all circuits associated with the switch, the lockout test and indication lights were tested and inspected with no problems found. The "trip exercised" light bulb had burned out which is a common occurrence.

The CM/CF System team will evaluate the testability features of the CFPT, evaluate the test procedure and make the necessary changes to improve the reliability of the test.

The DFS is designed to prevent a reactor trip in the event of the loss of one main feedwater pump. The DFS has been previously tested and proven able to handle the loss of one main feedwater pump without tripping the reactor. However, during this event the DFS was unable to prevent the reactor trip. The initial investigation of the DFS did not identify any problems; all systems and components appear to have functioned as designed. Therefore the root cause is unknown. Systems Engineering and Component Engineering will conduct further investigations to determine the root cause.

After the reactor trip the only anomalous behavior identified that affected plant performance was that the NC Loop B T-ave indication did not track down with the other loops. The T-ave indication began an erratic trend upward seconds after the CF isolation occurred. This resulted in the condenser steam dump valves opening to reduce the temperature in the NC system. Seconds later, T-ave in 2 of the 4 NC loops reached the P-12 interlock of 553 degrees F which closed all steam dump valves to prevent excessive cooldown of the NC system. The CROs monitoring NC system temperature throttled CA flow to control cooldown. The other Loop B temperature indications (hot leg and cold leg) were tracking with the other loops. The NC Loop B interlock was defeated and work order 9209510701 was issued to investigate and repair the T-ave indication. The investigation indicated a problem with the process control cards and all process control cards associated with Loop B T-ave were replaced and calibrated.

A review of the Operating Experience Program database indicates that there have been no incidents attributable to inadequate testability features during the past 24 months. Therefore this incident is not considered a recurring problem per Duke Power Company Safety Assurance guidelines.

CORRECTIVE ACTIONS

Immediate

1) CROs entered procedure AP/2/A/5500/03 (Load Rejection) for guidance and performed the required actions.

2) CROs entered procedure EP/2/A/5000/01 (Reactor Trip or Safety Injection) for guidance and performed the required actions.

3) Per guidance in EP/2/A/5000/01 (No Safety Injection occurred), CROs entered procedure EP/2/A/5000/01A (Reactor Trip Response) and performed the required actions.

4) CROs entered procedure AP/2/A/5500/12 (Loss of Charging or Letdown) for guidance and performed the required actions.

5) CROs throttled CA flow to control cooldown.

6) CROs defeated Delta-T and T-ave logic on NC B Loop.

7) Operators manually opened valves 2NV252 and 2NV253 in anticipation of an autoswap due to Volume Control Tank levels.

Subsequent

1) Work Order 9209478801 issued to determine the root cause of the CFPT 2A trip.

2) Work Order 9209510701 issued to investigate and repair the Unit 2 Loop B T-ave indications.

3) Systems Engineering (SES) investigated reactor trip and prepared reactor trip report.

4) SES evaluated the CA system response.

5) PIP 2-C92-0892 initiated to investigate why the Digital Feedwater System was unable to prevent the reactor trip.

Planned

1) Systems Engineering and Component Engineering will investigate why the Digital Feedwater System was unable to prevent the reactor trip.

2) CM/CF Systems Team will evaluate the testability features of the CFPTS, evaluate the test procedure, and make necessary changes to improve the reliability of the test.

3) Operations and Systems Engineering will determine if their test procedures (PTs) require manipulation of a similar spring loaded switch and if changes are necessary to prevent inadvertently tripping the component during testing.

SAFETY ANALYSIS

This incident was initiated on the loss of main feedwater pump 2A and is bounded by the Safety Analysis documented in the FSAR Section 15.2.7, Loss of Normal Feedwater Flow.

Upon loss of CFPT 2A, a pre-trip reactor and turbine run back was initiated. Steam Generator levels began decreasing and subsequently S/G 2D reached its lo lo level setpoint, resulting in reactor and turbine trip. All rods EIIS:ROD! inserted as expected to control reactivity. Auxiliary feedwater motor driven pumps also started, as expected, on the S/G lo lo level indication, to remove residual heat from the core. Subsequently, S/G

2C reached its lo lo level setpoint and the turbine driven auxiliary feedwater pump started on 2 of 4 S/G lo lo level indication. The average temperature (T-ave) of the Reactor Coolant System (NC) decreased below the CF isolation setpoint of 564 degrees F with the reactor tripped, and CF was automatically isolated to prevent excessive cooldown.

The NC Loop B T-ave indication did not operate as expected. The T-ave indication began an erratic trend upward seconds after the CF isolation occurred. This resulted in the condenser steam dump valves opening to reduce the temperature in the NC system. Seconds later, T-ave in 2 of 4 NC loops reached 553 degrees F and actuated the P-12 interlock, which closed all steam dump valves to prevent excessive cooldown of the NC system. The CROs monitoring the NC temperature throttled CA flow to control cooldown. The NC system reached a low

temperature of 530 degrees F, which is above the minimum temperature to ensure adequate shutdown margin. The cooldown of the NC system experienced in this incident is bounded by the Safety Analysis documented in the FSAR, Section 15.1.4, Inadvertent Opening of a Steam Generator Relief or Safety Valve.

After review of this incident all systems responded as designed, with the exception of the NC Loop B T-ave transmitter, to shutdown the Reactor and maintain it in a safe shutdown condition. Excessive heatup and overpressurization and excessive cooldown and underpressurization of the NC system was prevented. There were no actuations of Atmospheric Dumps, S/G PORVs, or S/G safety valves during this incident. There were no unusual releases of radioactive material.

The health and safety of the public were not affected by this incident.

ATTACHMENT 1 TO 9301250006 PAGE 1 OF 1

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DUKEPOWER

January 13, 1993

Document Control Desk
U. S. Nuclear Regulatory Commission

Washington, D.C. 20555

Subject: Catawba Nuclear Station
Docket No. 50-414
LER 414/92-006

Gentlemen:

Attached is Licensee Event Report 414/92-006 concerning MAIN FEEDWATER PUMP TRIP DURING TESTING AND SUBSEQUENT REACTOR TRIP.

This event was considered to be of no significance with respect to the health and safety of the public.

Very truly yours,

M. S Tuckman

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